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August 16, 2005

SVP-05-061

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Quad Cities Nuclear Power Station, Unit 1
Renewed Facility Operating License No. DPR-29
NRC Docket No. 50-254

Subject: Licensee Event Report 254/05-005, "Automatic Reactor Scram from High Reactor Pressure due to a Malfunction of the Electro-Hydraulic Control System"

Enclosed is Licensee Event Report (LER) 254/05-005, "Automatic Reactor Scram from High Reactor Pressure due to a Malfunction of the Electro-Hydraulic Control System," for Quad Cities Nuclear Power Station.

This report is submitted in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50.73 (a)(2)(iv)(A), which requires the reporting of any event or condition that resulted in manual or automatic actuation of the reactor protection system.

Should you have any questions concerning this report, please contact Mr. W. J. Beck at (309) 227-2800.

Respectfully,

TJ Tulon

Timothy J. Tulon
Site Vice President
Quad Cities Nuclear Power Station

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station

IE22

NRC FORM 366 (6-2004)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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4. TITLE Automatic Reactor Scram from High Reactor Pressure due to a Malfunction of the Electro-Hydraulic Control System

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									N/A	05000
06	17	2005	2005	- 005 -	00	08	16	2005	N/A	05000

9. OPERATING MODE: 1

10. POWER LEVEL: 085%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Wally Beck, Regulatory Assurance Manager	(309) 227-2800

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	JA	BD	G080	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

At 1120 hours (CDT) on June 17, 2005, the Unit 1 reactor automatically scrammed from 85% power due to a valid high reactor pressure signal. The maximum reactor pressure was approximately 1044 psig during the event. All control rods inserted to their full-in position. The reactor pressure increase was caused by a malfunction in the Electro-Hydraulic Control (EHC) system, which resulted in closure of the main turbine control valves. The main turbine bypass valves (nine) opened as expected in response to the pressure increase. No reactor pressure vessel safety or relief valves were required to actuate during the event. Reactor water level decreased to approximately -20 inches, which resulted in automatic Group 2 and 3 isolations as expected. All systems responded properly to the event.

The cause of the scram was a failure in one of the Control Valve Input Circuit Cards in the EHC system. Laboratory analysis was unable to identify the specific cause of failure. The circuit cards were replaced.

 PRINTED ON RECYCLED PAPER

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor, 2957 Megawatts Thermal Rated Core Power

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

EVENT IDENTIFICATION

Automatic Reactor Scram from High Reactor Pressure due to a Malfunction of the Electro-Hydraulic Control System

A. CONDITION PRIOR TO EVENT

Unit: 1	Event Date: June 17, 2005	Event Time: 1120 hours
Reactor Mode: 1	Mode Name: Power Operation	Power Level: 085%

Power Operation (1) - Mode switch in the RUN position with average reactor coolant temperature at any temperature.

B. DESCRIPTION OF EVENT

At 1120 hours (CDT) on June 17, 2005, the Unit 1 reactor automatically scrammed from 85% power due to a valid high reactor pressure signal. All control rods [AA] inserted to their full-in position. The maximum reactor pressure was approximately 1044 psig during the event. The reactor pressure increase was caused by a malfunction in the Electro-Hydraulic Control (EHC) system [JA], which resulted in closure of the main turbine control valves [TA]. The main turbine bypass valves (nine) opened as expected in response to the pressure increase. No reactor pressure vessel safety or relief valves [RV] were required to actuate during the event. Reactor water level decreased to approximately -20 inches, which resulted in automatic Group 2 and 3 isolations as expected. All systems responded properly to the event.

C. CAUSE OF EVENT

Based on information from instruments installed in response to previous anomalous indications, the cause of the scram was an intermittent failure in one of the Control Valve Input Circuit Cards in the EHC system. The Control Valve Amplifier, Load Limit, and Pressure Load Gate Amplifier cards, as well as their associated Operational Amplifier cards, were replaced, and the removed cards were subjected to failure analysis in a laboratory. The specific cause of the transient that initiated the system upset could not be determined. The root cause of the event is a transient malfunction of circuit cards within a non-fault tolerant EHC System (i.e., a transient anomaly that results in a system upset sufficient to cause a

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(If more space is required, use additional copies of NRC Form 366A)(17)

reactor scram without leaving any indication of its cause). No specific failure was identified.

D. SAFETY ANALYSIS

The safety significance of this event was minimal. All control rods inserted to their full in position, no reactor relief or safety valves were required to open, and all systems responded properly to the event.

E. CORRECTIVE ACTIONS

The Control Valve Amplifier, Load Limit, and Pressure Load Gate Amplifier cards, as well as their associated Operational Amplifier cards, were replaced, and the removed cards were subjected to failure analysis in a laboratory. No specific cause of failure was identified.

Quad Cities Nuclear Power Station is pursuing installation of a digital EHC system. This change will address the non-fault tolerant character of the system.

F. PREVIOUS OCCURRENCES

A review of LERs for the last three years did not identify any unit scrams due to EHC malfunction. However, since November 2002, there have been three instances of bad EHC cards and two instances of setpoint drift.

G. COMPONENT FAILURE DATA

The EHC cards are General Electric EHC Mark I Circuit Boards, component IDs 5600-A48, A49, A33, A34, A58 and A59.